December 28, 1995

Mr. J.P. O'Hanlon
Senior Vice President - Nuclear
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060

DISTRIBUTION:
See next page

SUBJECT: SURRY UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: SURRY, UNITS 1 AND 2 REACTOR VESSEL HEATUP AND COOLDOWN CURVES (TAC NOS. M92537 AND M92538)

Dear Mr. O'Hanlon:

The Commission has issued the enclosed Amendment No. 207 to Facility Operating License No. DPR-32 and Amendment No. 207 to Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application transmitted by letter dated June 8, 1995.

These amendments revise the reactor vessel pressure/temperature limits and an associated low temperature overpressure protective system setpoint that will be valid to the end-of-license (28.8 effective full power years (EFPY) and 29.4 EFPY for Surry, Units 1 and 2, respectively.

A copy of the Safety Evaluation is also enclosed. Also enclosed is the Notice of Issuance which has been forwarded to the Office of the Federal Register for publication.

Sincerely,
Original signed by:
Bart C. Buckley, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 207 to DPR-32
2. Amendment No. 207 to DPR-37
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures: See next page

CP-1

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Virginia Electric and Power Company

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DATED: December 28, 1995

AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-32 - SURRY UNIT 1
AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-37 - SURRY UNIT 2

Docket File
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 207
License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated June 8, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-32 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 207 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

DBMatthews

David B. Matthews, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 28, 1995

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 207
License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated June 8, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-37 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 207 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

D B Matthews

David B. Matthews, Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 28, 1995

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-32

AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NOS. 50-280 AND 50-281

Revise Appendix A as follows:

Remove Pages	Insert Pages
TS 3.1-3	TS 3.1-3
TS 3.1-6	TS 3.1-6
TS 3.1-9	TS 3.1-9
TS 3.1-10	TS 3.1-10
TS 3.1-11	TS 3.1-11
TS 3.1-12	TS 3.1-12
TS 3.1-18	TS 3.1-18
TS 3.1-19	TS 3.1-19
TS 3.1-23a	TS 3.1.23a
TS 3.1-26	TS 3.1-26
TS 3.1-27	TS 3.1-27
TS 3.1-28	TS 3.1-28
TS 3.1-29	TS 3.1-29
Figure 3.1-1	Figure 3.1-1
Figure 3.1-2	Figure 3.1-2
Figures 3.1-3 and 3.1-4	---

e. When all three pumps have been idle for > 15 minutes, the first pump shall not be started unless: (1) a bubble exists in the pressurizer or (2) the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

2. Steam Generator

A minimum of two steam generators in non-isolated loops shall be OPERABLE when the average Reactor Coolant System temperature is greater than 350°F.

3. Pressurizer Safety Valves

a. Three valves shall be OPERABLE when the head is on the reactor vessel and the Reactor Coolant System average temperature is greater than 350°F, the reactor is critical, or the Reactor Coolant System is not connected to the Residual Heat Removal System.

b. Valve lift settings shall be maintained at 2485 psig ± 1 percent*

---

* The as-found tolerance shall be ±3% and the as-left tolerance shall be ±1%.

B. HEATUP AND COOLDOWN

Specification

1. Unit 1 and Unit 2 reactor coolant temperature and pressure and the system heatup and cooldown (with the exception of the pressurizer) shall be limited in accordance with TS Figures 3.1-1 and 3.1-2.

   Heatup:

   Figure 3.1-1 may be used for heatup rates of up to 60°F/hr.

   Cooldown:

   Allowable combinations of pressure and temperature for specific cooldown rates are below and to the right of the limit lines as shown in TS Figure 3.1-2. This rate shall not exceed 100°F/hr. Cooldown rates between those shown can be obtained by interpolation between the curves on Figure 3.1-2.

   Core Operation:

   During operation where the reactor core is in a critical condition (except for low level physics tests), vessel metal and fluid temperature shall be maintained above the reactor core criticality limits specified in 10 CFR 50 Appendix G. The reactor shall not be made critical when the reactor coolant temperature is below 522°F as specified in T.S. 3.1.E.

2. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 70°F.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 28.8 Effective Full Power Years (EFPY) and 29.4 EFPY for Units 1 and 2, respectively. The most limiting value of RT_{NDT} (228.4°F) occurs at the 1/4-T, 0° azimuthal location in the Unit 1 intermediate-to-lower shell circumferential weld. The limiting RT_{NDT} at the 1/4-T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. This ensures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are presented in report BAW-2222, "Response to Closure Letters to NRC Generic Letter 92-01, Revision 1," dated June, 1994 and are reproduced in Tables 3.1-1 and 3.1-2. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the copper and nickel content of the material and the fluence was calculated in accordance with the recommendations of Regulatory Guide 1.99, Revision 2 "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.1-1 and 3.1-2 include predicted adjustments for this shift in RT_{NDT} at the end of 28.8 EFPY and 29.4 EFPY for Units 1 and 2, respectively (as well as adjustments for location of the pressure sensing instrument).

Surveillance capsules will be removed in accordance with the requirements of ASTM E185-82 and 10 CFR 50, Appendix H. The surveillance specimen withdrawal schedule is shown in the UFSAR. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure, or when the service period exceeds 28.8 EFPY or 29.4 EFPY for Units 1 and 2, respectively, prior to a scheduled refueling outage.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of one and one half T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

where K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\, K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

where, K_{IM} is the stress intensity factor caused by membrance (pressure) stress.

K_{It} is the stress intensity factor caused by the thermal gradients

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material.

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{It}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

The heatup limit curve, Figure 3.1-1, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves of Figure 3.1-2 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The cooldown limit curves are valid for cooldown rates up to 100°F/hr. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 28.8 EFPY and 29.4 EFPY for Units 1 and 2, respectively. The adjusted reference temperature was calculated using materials properties data from the B&W Owners Group Master Integrated Reactor Vessel Surveillance Program (MIRVSP) documented in the most recent revision to BAW-1543 and reactor vessel neutron fluence data obtained from plant-specific analyses.

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E. Minimum Temperature for Criticality

Specifications

1. Except during LOW POWER PHYSICS TESTS, the reactor shall not be made critical at any Reactor Coolant System temperature above which the moderator temperature coefficient is more positive than the limit specified in the CORE OPERATING LIMITS REPORT. The maximum upper limit for the moderator temperature coefficient shall be:

   a. + 6 pcm/°F at less than 50% of RATED POWER, or
   b. + 6 pcm/°F at 50% of RATED POWER and linearly decreasing to 0 pcm/°F at RATED POWER.

2. In no case shall the reactor be made critical with the Reactor Coolant System temperature below the limiting value of RT_{NDT} + 10°F, where the limiting value of RT_{NDT} is as determined in Part B of this specification.

3. When the Reactor Coolant System temperature is below the minimum temperature as specified in E-2 above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to primary coolant depressurization.

4. The reactor shall not be made critical when the Reactor Coolant System temperature is below 522°F.

Basis

During the early part of a fuel cycle, the moderator temperature coefficient may be calculated to be slightly positive at coolant temperatures in the power operating range. The moderator coefficient will be most positive at the beginning of cycle life, when the boron concentration in the coolant is the greatest. Later in the cycle, the boron concentration in the coolant will be lower and the moderator coefficient will be less positive or will be negative in the power operating range. At the beginning of cycle life, during pre-operational physics tests, measurements are made to determine that the moderator coefficient is less than the limit specified in the CORE OPERATING LIMITS REPORT.

The requirement that the reactor is not to be made critical when the moderator coefficient is greater than the low power limit specified in the CORE OPERATING LIMITS REPORT has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant pressure. This requirement is waived during LOW POWER PHYSICS TESTS to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operation precautions will be taken. In addition, the strong negative Doppler coefficient (2)(3) and the small integrated Delta k/k would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical with a Reactor Coolant System temperature below the limiting value of RT_{NDT} + 10°F provides increased assurance that the proper relationship between Reactor Coolant System pressure and temperature will be maintained during system heatup and pressurization whenever the reactor vessel is in the nil ductility transition temperature range. Heatup to this temperature is accomplished by operating the reactor coolant pumps.

The requirement that the reactor is not to be made critical with a Reactor Coolant System temperature below 522°F provides added assurance that the assumptions made in the safety analyses remain bounding by maintaining the moderator temperature within the range of those analyses.

If a specified shutdown reactivity margin is maintained (TS Section 3.12), there is no possibility of an accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.

(1) UFSAR Figure 3.3-8
(2) UFSAR Table 3.3-1
(3) UFSAR Figure 3.3-9

(3) During the initial 72 hours, maintain a bubble in the pressurizer with a maximum narrow range level of 33%,

or

(4) Maintain two Power Operated Relief Valves (PORV) OPERABLE with a lift setting of ≤ 390 psig and verify each PORV block valve is open at least once per 72 hours,

or

(5) The RCS shall be vented through one open PORV or an equivalent size opening as specified below:

(a) with the RCS vented through an unlocked open vent path, verify the path is open at least once per 12 hours, or

(b) with the RCS vented through a locked open vent path verify the path is open at least once per 31 days.

2. The requirements of Specification 3.1.G.1.c.(4) may be modified as follows:

a One PORV may be inoperable in INTERMEDIATE SHUTDOWN with the RCS average temperature > 200°F but < 350°F for a period not to exceed 7 days. If the inoperable PORV is not restored to OPERABLE status within 7 days, then completely depressurize the RCS and vent through one open PORV or an equivalent size opening within the next 8 hours.

b One PORV may be inoperable in COLD SHUTDOWN or REFUELING SHUTDOWN with the reactor vessel head bolted for a period not to exceed 24 hours. If the inoperable PORV is not restored to OPERABLE status within 24 hours then completely depressurize the RCS and vent through one open PORV or an equivalent size opening within 8 hours.

TABLE 3.1-1

UNIT 1 REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)[d]

MATERIAL	HEAT OR CODE NO.	MATERIAL SPEC. NO.	Cu (%)	Ni (%)	T_{NDT} (°F)	RT_{NDT} (°F)	NMWD[b] UPPER SHELF ENERGY (FT LB)
Closure head dome	C4315-2	A53313 Cl. 1	.14	.59	0	0	75
Head flange	FV-1894	A508 Cl. 2	.13	.64	10[a]	10	125
Vessel flange	FV-1870	A508 Cl. 2	.10	.65	10[a]	10	74
Inlet nozzle	9-5078	A508 Cl. 2	-	.87	60[a]	60	64
Inlet nozzle	9-4819	A508 Cl. 2	-	.84	60[a]	60	68
Inlet nozzle	9-4787	A508 Cl. 2	-	.85	60[a]	60	64
Outlet nozzle	9-4762	A508 Cl. 2	-	.83	60[a]	60	85
Outlet nozzle	9-4788	A508 Cl. 2	-	.84	60[a]	60	72
Outlet nozzle	9-4825	A508 Cl. 2	-	.85	60[a]	60	68
Upper shell	122V109	A508 Cl. 2	.09	.74	40	40	83
Intermediate shell	C4326-1	A533B Cl. 1	.11	.55	10	10	115[c]
Intermediate shell	C4326-2	A533B Cl. 1	.11	.55	0	0	94
Lower shell	C4415-1	A533B Cl. 1	.11	.50	20	20	103[c]
Lower shell	C4415-2	A533B Cl. 1	.11	.50	0	0	83
Bottom head ring	123T338	A508 Cl. 2	-	.69	50	50	86
Bottom dome	C4315-3	A533B Cl. 1	.14	.59	0	0	85
Inter. & lower shell vertical weld seam L1, L3, & L4	8T1554 & Linde 80 flux		.18	.63	0[a]	-5	77/EMA[e]

TABLE 3.1-1 (Continued)

UNIT 1 REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)[d]

MATERIAL	HEAT OR CODE NO.	MATERIAL SPEC. NO.	Cu (%)	Ni (%)	T_{NDT} (°F)	RT_{NDT} (°F)	NMWD[b] UPPER SHELF ENERGY (FT LB)
Lower shell vertical weld seam, L2	299L44 & Linde 80 flux		.35	.68	0[a]	-7	70/EMA[e]
Inter. to lower shell girth seam	72445 & Linde 80 flux		.21	.59	0[a]	-5	77[a]/EMA[e]
Upper shell to Inter. shell girth seam	25017 & SAF 89 flux		.33	.10	0[a]	0	EMA[e]

NOTES:

(a) Estimated per NRC standard review plan, NUREG-0800, Section MTEB 5-2

(b) Normal to major working direction - estimated per NRC standard review plan, NUREG-0800, Section MTEB 5-2

(c) Actual values

(d) Reactor Vessel Fabricator Certified Test Reports

(e) The approved equivalent margins analysis in the Topical Reports BAW-2192PA and BAW-2178PA demonstrates compliance with the requirements of 10 CFR 50, Appendix G.

TABLE 3.1-2

UNIT 2 REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

MATERIAL	HEAT OR CODE NO.	MATERIAL SPEC. NO.	Cu (%)	Ni (%)	T_{NDT} (°F)	RT_{NDT} (°F)	NMWD[b] UPPER SHELF ENERGY (FT LB)
Closure head dome	C4361-2	A533B Cl. 1	.15	.52	-20	7	81
Head flange	ZV-3475	A508 Cl. 2	.11	.60	<10[a]	<10	129
Vessel flange	ZV-3476	A508 Cl. 2	.10	.64	-65[a]	-65	129
Inlet nozzle	9-4815	A508 Cl. 2	-	.87	60[a]	60	66
Inlet nozzle	9-5104	A508 Cl. 2	-	.84	60[a]	60	73
Inlet nozzle	9-5205	A508 Cl. 2	-	.86	60[a]	60	66
Outlet nozzle	9-4825	A508 Cl. 2	-	.85	60[a]	60	74
Outlet nozzle	9-5086	A508 Cl. 2	-	.86	60[a]	60	79
Outlet nozzle	9-5086	A508 Cl. 2	-	.87	60[a]	60	73
Upper shell	123V303	A508 Cl. 2	.09	.73	30	30	104
Intermediate shell	C4331-2	A533B Cl. 1	.12	.60	-10	-10	84
Intermediate shell	C4339-2	A533B Cl. 1	.11	.54	-20	-20	83
Lower shell	C4208-2	A533B Cl. 1	.15	.55	-30	-30	94
Lower shell	C4339-1	A533B Cl. 1	.11	.54	-10	-10	105[c]
Bottom head ring	123T321	A508 Cl. 2	-	.71	10	10	101
Bottom dome	C4361-3	A533B Cl. 1	.15	.52	-20	-15	80
Intermediate shell vertical weld seams L3 (100%), L4 (OD50%)	72445 & Linde 80 flux Lot 8579		.21	.59	-	-5	77[a]/EMA[d]
L4 (ID50%)	8T1762 & Linde 80 flux 8597		.20	.55	-	-5	EMA[d]

TABLE 3.1-2 (Continued)

UNIT 2 REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

MATERIAL	HEAT OR CODE NO.	MATERIAL SPEC. NO.	Cu (%)	Ni (%)	T_{NDT} (°F)	RT_{NDT} (°F)	NMWD[b] UPPER SHELF ENERGY (FT LB)
Lower shell vertical welds							
Seam L2 (ID 63%)	8T1762 & Linde 80 flux 8597		.20	.55	-	-5	EMA[d]
Seam L1 (100%)	8T1762 & Linde 80 flux 8597		.20	.55	-	-5	EMA[d]
Seam L2 (OD37%)	8T1762 & Linde 80 flux 8632		.20	.55	-	-5	EMA[d]
Inter. to lower shell girth seam	0227 and Grau Lo Flux LW320		.19	.56	0[a]	0	90[c]/EMA[d]
Upper shell to Inter. shell girth seam	4275 & SAF 89 flux		.35	.10	0[a]	0	EMA[d]

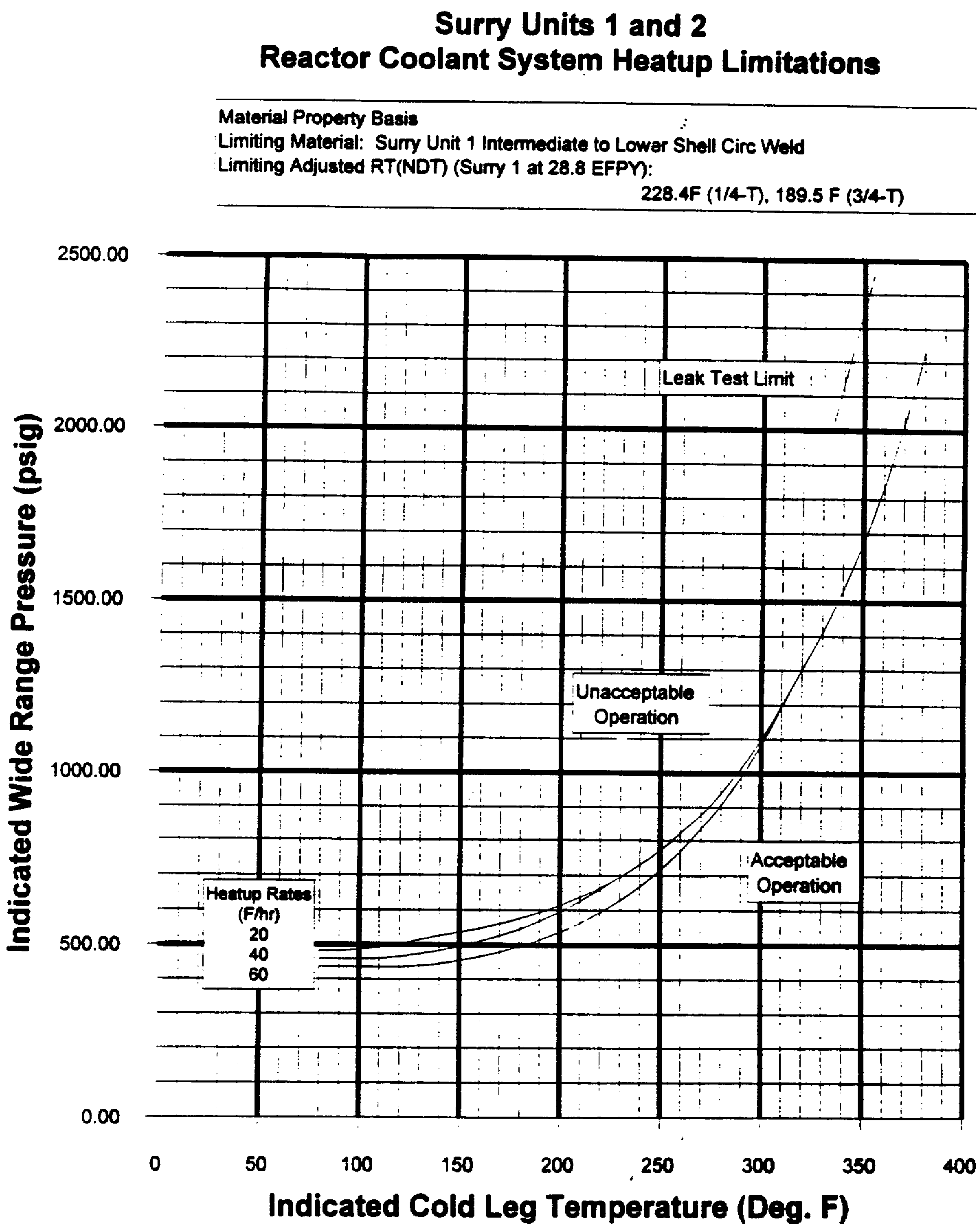
NOTES:

(a) Estimated per NRC standard review plan, NUREG-0800, Section MTEB 5-2

(b) Normal to major working direction - estimated per NRC standard review plan, NUREG-0800, Section MTEB 5-2

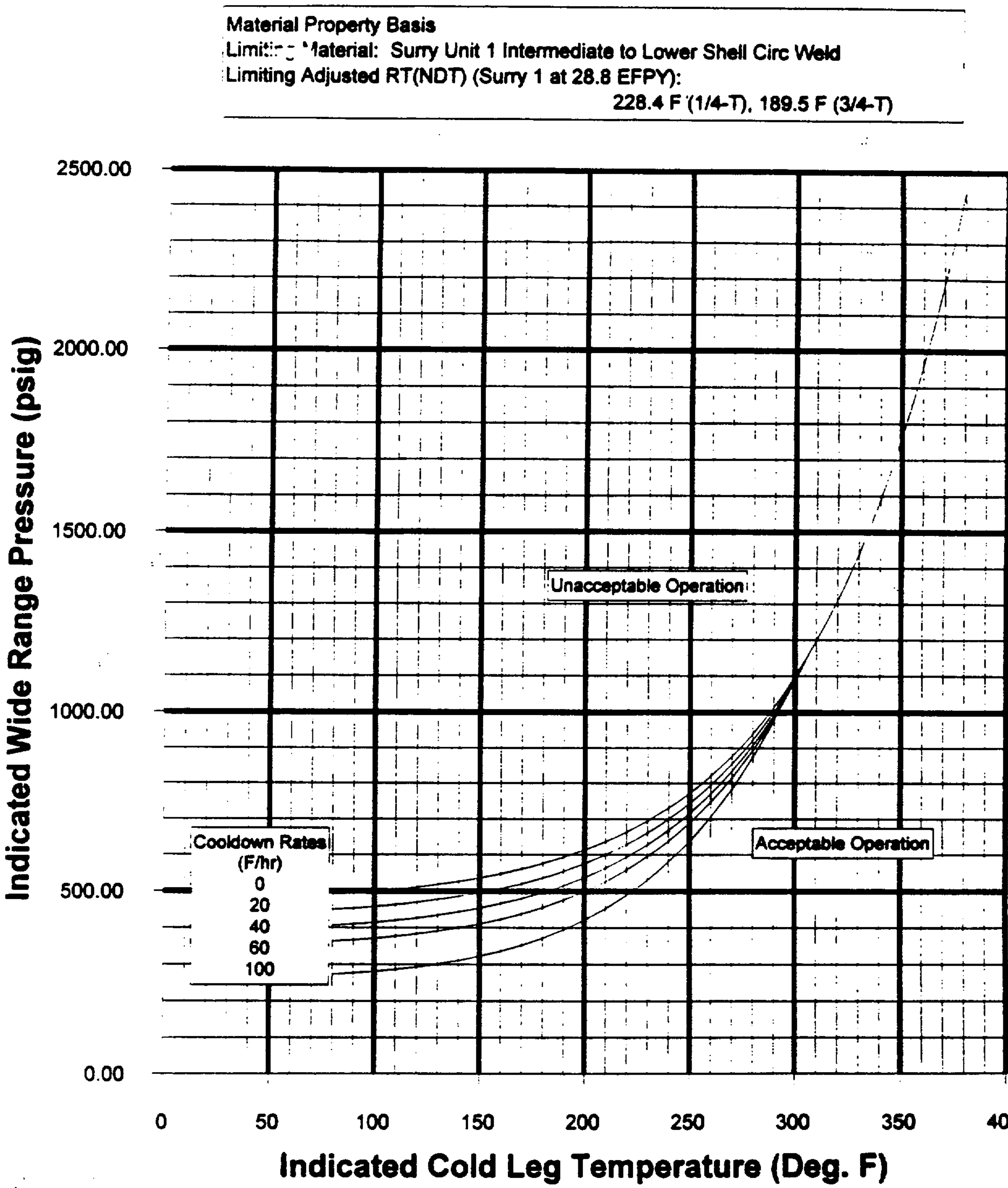
(c) Actual value based on surveillance tests normal to the major working direction

(d) The approved equivalent margins analysis in the Topical Reports BAW-2192PA and BAW-2178PA demonstrates compliance with the requirements of 10 CFR 50, Appendix G.



Surry Units 1 and 2 Reactor Coolant System Heatup Limitations (Heatup Rates up to 60 F/hr) Applicable for the First 28.8 EFPY for Surry Unit 1 and the First 29.4 EFPY for Surry Unit 2

Surry Units 1 and 2
Reactor Coolant System Cooldown Limitations



Surry Units 1 and 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100 F/hr) Applicable for the First 28.8 EFPY for Surry Unit 1 and the First 29.4 EFPY for Surry Unit 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-32

AND AMENDMENT NO. 207 TO FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

By letter dated June 8, 1995, the Virginia Electric and Power Company (the licensee) submitted changes to the pressure-temperature (P/T) limits in the Surry Units 1 and 2 Technical Specifications (TS). The licensee revised the P/T limits to provide new limits that are valid to the end-of-license (28.8 effective full power years (EFPY) for Unit 1 and 29.4 EFPY for Unit 2). The licensee also proposed a revision to the associated low temperature overpressure protection (LTOP) enabling temperature methodology and the power operated relief valve (PORV) setpoint.

The staff evaluated the P/T limits based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; Generic Letters 88-11 and 92-01; Regulatory Guide (RG) 1.99, Rev. 2; and Standard Review Plan (SRP) Section 5.3.2. Appendix G to 10 CFR Part 50 requires that P/T limits for the reactor vessel must be at least as conservative as those obtained by Appendix G to Section III of the ASME Code. GL 88-11 requires that licensees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation by calculating the adjusted reference temperature (ART) of reactor vessel materials. The ART is defined as the sum of the initial nil-ductility transition reference temperature (RT_{NDT}) of the material, the increase in RT_{NDT} caused by neutron irradiation, and a margin to account for uncertainties in the prediction method. The increase in RT_{NDT} is calculated from the product of a chemistry factor and a fluence factor. The chemistry factor may be calculated using credible surveillance data, obtained by the licensee's surveillance program, as directed by Position 2 of Regulatory Guide (RG) 1.99, Rev. 2. If credible surveillance data is not available, the chemistry factor is calculated dependent upon the amount of copper and nickel in the vessel material as specified in Table 1 of RG 1.99, Rev. 2. GL 92-01 requires licensees to submit reactor vessel materials data, which the staff uses in the review of the P/T limits submittals.

SRP 5.3.2 provides guidance on calculation of the P/T limits using linear elastic fracture mechanics methodology specified in Appendix G to Section III of the ASME Code. The linear elastic fracture mechanics methodology postulates sharp surface defects that are normal to the direction of maximum stress and have a depth of one-fourth of the reactor vessel beltline thickness (1/4T) and a length of 1-1/2 times the beltline thickness. The critical locations in the vessel for this methodology are the 1/4T and 3/4T locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

2.0 EVALUATION

2.1 Pressure Vessel Heatup and Cooldown Curves

For the Surry Units 1 and 2 reactor vessels, the licensee determined that the most limiting materials at the 1/4T and 3/4T locations is the intermediate-to-lower shell circumferential weld (SA-1585) in Unit 1. The licensee calculated an ART of 228.4°F at the 1/4T location and an ART of 189.5°F at the 3/4T location at 28.8 EFPY. The integrated surveillance data used for this material is based on B&WOG Capsule CR3-LG1 and Point Beach 1 surveillance data. The neutron fluence used in the ART calculation was 2.45×10^{19} n/cm^2 at the 1/4T location and 0.938×10^{19} n/cm^2 at the 3/4T location. The initial RT_{NDT} for the limiting weld was -5°F, the generic value reported in BAW-2166. The margin term used in calculating the ART for the limiting weld was 48°F.

The staff verified for Surry Units 1 and 2 that the copper and nickel content and initial RT_{NDT} agreed with the reactor vessel integrity database (RVID) as reported by the licensee in response to GL 92-01. The staff used the data from the RVID to perform an independent calculation of the ART values for the limiting materials using RG 1.99, Revision 2. The staff also verified that the licensee's surveillance data meet the credibility criteria of the RG. Evaluation of criteria 3 and 5 which address scatter in ΔRT_{NDT} values and correlation monitor materials, respectively will be presented in more detail in the following paragraph. In addition, the staff used the surveillance data, as submitted in previous reports to the NRC, to perform an independent calculation of the ART values for the surveillance materials using Position 2 of RG 1.99, Revision 2.

Comparisons of measured to predicted ΔRT_{NDT} values for all surveillance material are shown in Tables 1 and 2 for Units 1 and 2, respectively. The comparison of the measured to predicted ΔRT_{NDT} values indicate that all measured values are within one standard deviation (17°F for plates and 28°F for welds) of the predicted values which satisfies credibility criterion 3. Figure 1 shows ΔRT_{NDT} values vs. fluence for all correlation monitor materials from heat SHSS02 (heat designation in the Power Reactor Embrittlement Database (PREDB). The material is from the Oak Ridge National Laboratory (ORNL) Heavy Section Steel Technology (HSST) program. The plot indicates that the surveillance data for the Surry Units 1 and 2 correlation monitor materials are within the scatter band of the entire material data base which satisfies credibility criterion 5.

Based on the calculations, the staff verified that the licensee's limiting material for Surry Units 1 and 2 is the intermediate-to-lower shell circumferential weld (SA-1585) in Unit 1. The staff's calculated ART values for the limiting materials agreed with the licensee's calculated ART values.

Substituting the ART values for Surry Units 1 and 2 into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown and hydrostatic tests satisfy the requirements in Paragraphs IV.A.2 and IV.A.3 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange RT_{NDT} of 10°F for Unit 1 and <10 °F for Unit 2 provided by the licensee, the staff has determined that the proposed P/T limits have satisfied the requirement for the closure flange region during normal operation, hydrostatic pressure test and leak test.

2.2 Enabling Temperature

The enabling temperature is presently set at 350 ^{0}F and no change has been proposed. The reason is that 350 ^{0}F is conservative using the ASME Code Case N-514 required calculation of the enabling temperature i.e. RT_{NDT} + 50 ^{0}F + (instrument uncertainty). The new RT_{NDT} value has been estimated at 228.4 ^{0}F and instrument uncertainty at 21 ^{0}F for an enabling temperature of 299.4 ^{0}F. The existing enabling temperature of 350 ^{0}F is conservative and, therefore, is acceptable.

2.3 PORV Lift Setpoint.

A PORV lift setpoint of ≤ 390 psig has been proposed. To validate this value the licensee estimated the pressure overshoot in a mass addition (inadvertent startup of a charging pump) and heat addition transients (inadvertent reactor coolant system (RCS) pump startup with a 50 ^{0}F ΔT between the RCS and the steam generators). The temperatures and corresponding estimated pressures are as follows: (100 ^{0}F, 200 psig), (150 ^{0}F, 250 psig), (200 ^{0}F, 300 psig), (250 ^{0}F,340 psig), (300 ^{0}F, 380 psig), (325 ^{0}F, 400 psig). Therefore, for a 350 ^{0}F enabling temperature the proposed 390 psig is a conservative lift setpoint and it is acceptable.

3.0 SUMMARY

The staff has performed an independent analysis to verify the licensee's proposed P/T limits. The staff concludes that the proposed P/T limits for heatup, cooldown and hydrostatic tests are valid until the end-of-license because 1) the limits conform to the requirements of Appendix G of 10 CFR Part 50 and GL 88-11, 2) the material properties and chemistry used in calculating the P/T limits are consistent with data submitted under GL 92-01, 3) the surveillance data used in calculating the P/T limits are consistent with data in surveillance reports submitted to the staff, and 4) the surveillance data meets the credibility criteria of RG 1.99, Rev. 2. Hence, the proposed P/T limits may be incorporated in the Surry Units 1 and 2 Technical Specifications. In addition, the proposed editorial changes in the Bases section of the Technical Specifications are consistent with the P/T limits change; therefore, they are acceptable. Moreover, based upon our review of the revised enabling temperature methodology, we find the 350°F enabling temperature and the proposed PORV setpoint of 390 psig acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendments. The State official had no comment.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.32 an environmental assessment has been published (60 FR 54710) in the Federal Register on October 25, 1995. Accordingly, the Commission has determined that the issuance of this amendment will not result in any environmental impacts other than those evaluated in the Final Environmental Statement.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988

2. NUREG-0800, Standard Review Plan, Section 5.3.2: Pressure-Temperature Limits

3. Code of Federal Regulations, Title 10, Part 50, Appendix G, Fracture Toughness Requirements

4. Generic Letter 88-11, NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations, July 12, 1988

5. ASME Boiler and Pressure Vessel Code, Section III, Appendix G for Nuclear Power Plant Components, Division 1, "Protection Against Nonductile Failure"

6. June 8, 1995, Letter from J. P. O'Hanlon to USNRC Document Control Desk, Subject: Request for Exemption - ASME Code Case N-514 Proposed Technical Specifications Change Revised Pressure/Temperature Limits and LTOPS Setpoint

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Date: December 28, 1995

COMPARISON OF MEASURED TO PREDICTED DELTA RTndt FOR SURRY UNIT 1 SURVEILLANCE MATERIAL

MATERIAL	CAPSULE	Δ RTndt (Measured)	Δ RTndt (Predicted)	Measured-Predicted
Lower Shell Plate C4415-1	T	50	56	-6
	V	110	107	3
Lower Shell Axial Weld L2 SA-1526	TMI 1 - E (WF-25)	124	119	5
	TMI 1 - C (WF-25)	203	213	-10
	B&WOG CR3-LG1 (WF25)	214	191	23
	V	240	262	-22
	T	167	146	21
Intermediate to Lower Shell Circ. Weld SA-1585	B&WOG CR3-LG1	148	132	16
	Point Beach 1 - V (SA-1263)	110	109	1
	Point Beach 1 - S (SA-1263)	165	142	23
	Point Beach 1 - R (SA-1263)	165	186	-21
	Point Beach 1 - T (SA-1263)	175	184	-9

TABLE 1

COMPARISON OF MEASURED TO PREDICTED DELTA RTndt FOR SURRY UNIT 2 SURVEILLANCE MATERIAL

MATERIAL	CAPSULE	Δ RTndt (Measured)	Δ RTndt (Predicted)	Measured-Predicted
Lower Shell Plate	X	50	45	5
C4339-1	V	75	78	-3
Intermediate to Lower Shell Circ. Weld	X	95	86	9
R3008	V	145	150	-5

TABLE 2

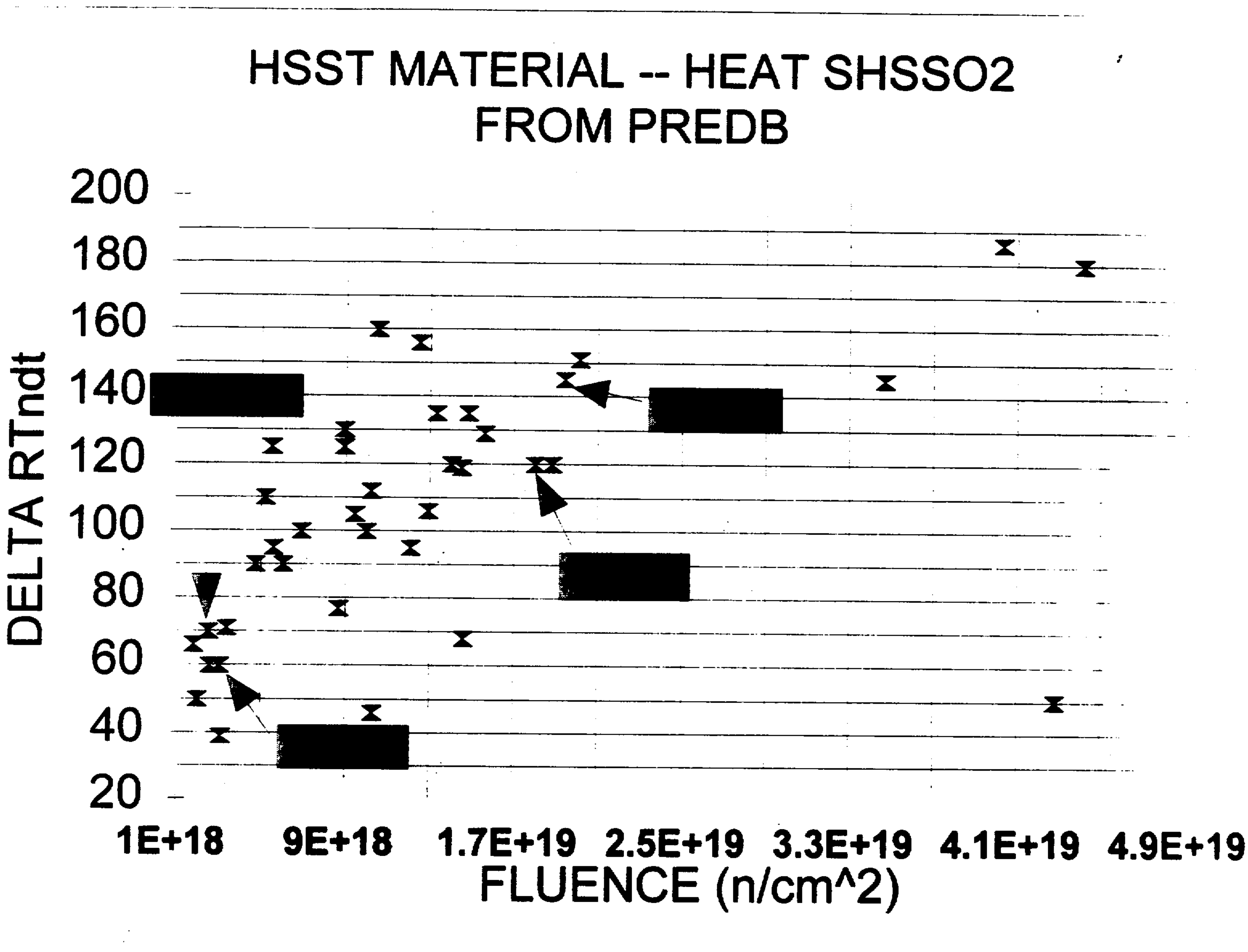
HSST MATERIAL -- HEAT SHSSO2
FROM PREDB
DELTA RTndt
200
180
160
140
120
100
80
60
40
20
1E+18
9E+18
1.7E+19
2.5E+19
3.3E+19
4.1E+19
4.9E+19
FLUENCE (n/cm^2)

FIGURE 1

UNITED STATES NUCLEAR REGULATORY COMMISSION

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NOS. 50-280 AND 50-281

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment Nos. 207 and 207 to Facility Operating License Nos. DPR-32 and DPR-37 issued to the Virginia Electric and Power Company (the licensee), which revised the Technical Specifications for operation of the Surry Power Station, Units 1 and 2 located in Surry County, Virginia. The amendments are effective as of the date of issuance.

The amendments modified the Technical Specifications to revise the reactor vessel pressure/temperature limit and an associated low temperature overpressure protection system setpoint.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments.

Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing in connection with this action was published in the FEDERAL REGISTER on August 4, 1995 (60 FR 39978). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendments will not have a significant effect on the quality of the human environment (60 FR 54710).

For further details with respect to the action see (1) the application for amendments dated June 8, 1995, (2) Amendment Nos. 207 and 207 to License Nos. DPR-32 and DPR-37, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC, and at the local public document room located at the Swem Library, College of William and Mary, Williamsburg, Virginia 23185.

Dated at Rockville, Maryland, this 28th day of December, 1995.

FOR THE NUCLEAR REGULATORY COMMISSION

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